November 26, 2002

Mr. Fred J. Cayia
Site Vice President
Point Beach Nuclear Plant
Nuclear Management Company, LLC
6610 Nuclear Road
Two Rivers, WI 54241

SUBJECT: POINT BEACH NUCLEAR PLANT UNITS 1 AND 2 - ISSUANCE OF AMENDMENTS RE: CHANGE OF CONTAINMENT MAXIMUM PRESSURE TECHNICAL SPECIFICATION LIMIT (TAC NOS. MB3870 AND MB3871)

Dear Mr. Cayia:

The Commission has issued the enclosed Amendment No. 206 to Facility Operating License No. DPR-24 and Amendment No. 211 to Facility Operating License No. DPR-27 for the Point Beach Nuclear Plant, Units 1 and 2, respectively. The amendments consist of changes to the Technical Specifications (TSs) in response to your application dated January 11, 2002, as supplemented by letters dated August 19 and September 12, 2002.

These amendments revise TS 3.6.4, "Containment Pressure," to reduce the maximum allowable pressure from 3 pounds per square inch gauge (psig) to 2 psig. This change is necessary to support calculations of a postulated main steamline break accident with a worst single failure not previously considered

A copy of our related safety evaluation is also enclosed. The Notice of Issuance will be included in the Commission's biweekly *Federal Register* notice.

Sincerely,

/RA/

Deirdre W. Spaulding, Project Manager, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Docket Nos. 50-266 and 50-301

Enclosures:
1. Amendment No. 206 to DPR-24
2. Amendment No. 211 to DPR-27
3. Safety Evaluation

cc w/encls: See next page

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cc w/encls: See next page

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**See previous concurrence
*Provided SE input by memo

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NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-266

POINT BEACH NUCLEAR PLANT, UNIT 1

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 206
License No. DPR-24

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated January 11, 2002, as supplemented August 19 and September 12, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-24 is hereby amended to read as follows:

   B. Technical Specifications

      The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 206, are hereby incorporated in the license. The licensee shall operate the facility in accordance with Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 45 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: November 26, 2002

NUCLEAR MANAGEMENT COMPANY, LLC

DOCKET NO. 50-301

POINT BEACH NUCLEAR PLANT, UNIT 2

AMENDMENT TO FACILITY OPERATING LICENSE

Amendment No. 211
License No. DPR-27

1. The Nuclear Regulatory Commission (the Commission) has found that:

   A. The application for amendment by Nuclear Management Company, LLC (the licensee), dated January 11, 2002, as supplemented August 19 and September 12, 2002, complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations set forth in 10 CFR Chapter I;

   B. The facility will operate in conformity with the application, the provisions of the Act, and the rules and regulations of the Commission;

   C. There is reasonable assurance (i) that the activities authorized by this amendment can be conducted without endangering the health and safety of the public, and (ii) that such activities will be conducted in compliance with the Commission's regulations;

   D. The issuance of this amendment will not be inimical to the common defense and security or to the health and safety of the public; and

   E. The issuance of this amendment is in accordance with 10 CFR Part 51 of the Commission's regulations and all applicable requirements have been satisfied.

2. Accordingly, the license is amended by changes to the Technical Specifications as indicated in the attachment to this license amendment, and paragraph 3.B of Facility Operating License No. DPR-27 is hereby amended to read as follows:

    B. Technical Specifications

    The Technical Specifications contained in Appendices A and B, as revised through Amendment No. 211, are hereby incorporated in the license. The licensee shall operate the facility in accordance with Technical Specifications.

3. This license amendment is effective as of the date of issuance and shall be implemented within 45 days of the date of issuance.

FOR THE NUCLEAR REGULATORY COMMISSION

/RA/

L. Raghavan, Chief, Section 1
Project Directorate III
Division of Licensing Project Management
Office of Nuclear Reactor Regulation

Attachment: Changes to the Technical Specifications

Date of issuance: November 26, 2002

ATTACHMENT TO LICENSE AMENDMENT NO. 206

TO FACILITY OPERATING LICENSE NO. DPR-24

AND LICENSE AMENDMENT NO. 211

TO FACILITY OPERATING LICENSE NO. DPR-27

DOCKET NOS. 50-266 AND 50-301

Replace the following pages of the Appendix A Technical Specifications with the attached revised pages. The revised pages are identified by amendment number and contain marginal lines indicating the areas of change.

REMOVE	INSERT
3.6.4-1	3.6.4-1
B 3.6.4-1	B 3.6.4-1
B 3.6.4-2	B 3.6.4-2
B 3.6.4-3	B 3.6.4-3
B 3.6.4-4	-
B 3.6.5-1	B 3.6.5-1
B 3.6.5-2	B 3.6.5-2

SAFETY EVALUATION BY THE OFFICE OF NUCLEAR REACTOR REGULATION

RELATED TO AMENDMENT NO. 206 TO FACILITY OPERATING LICENSE NO. DPR-24

AND AMENDMENT NO. 211 TO FACILITY OPERATING LICENSE NO. DPR-27

NUCLEAR MANAGEMENT COMPANY, LLC

POINT BEACH NUCLEAR PLANT, UNITS 1 AND 2

DOCKET NOS. 50-266 AND 50-301

1.0 INTRODUCTION

By application dated January 11, 2002, as supplemented by letters dated August 19 and September 12, 2002, the Nuclear Management Company, LLC (the licensee), requested changes to the Technical Specifications (TSs) for the Point Beach Nuclear Plant, Units 1 and 2. The August 19 and September 12, 2002, supplemental letters provided additional information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the Nuclear Regulatory Commission (NRC) staff's original proposed no significant hazards consideration determination as published in the *Federal Register* on March 19, 2002 (67 FR 12605).

The proposed changes would revise TS 3.6.4, "Containment Pressure," to reduce the maximum allowable pressure from 3 pounds per square inch gauge (psig) to 2 psig. Specifically, Limiting Condition for Operation (LCO) 3.6.4 would be revised to state:

> "Containment pressure shall be ≥ -2.0 psig and ≤ +2.0 psig."

The proposed change to LCO 3.6.4, "Containment Pressure," to decrease the allowable maximum containment internal pressure from ≤3.0 psig to ≤2.0 psig would be applicable in Modes 1, 2, 3, and 4. These are modes in which the reactor coolant system and steam generators are pressurized and the containment is required to be operable. This change is required as a result of a reanalysis of the postulated main steamline break accident to address a single failure not previously considered. This single failure is the postulated failure of a feedwater regulating valve to close when required during a postulated steamline break at full power. The failure of a feedwater regulating valve to close is the worst single failure. LCO 3.6.4 also specifies a minimum containment pressure of ≥ -2.0 psig during operation in Modes 1, 2, 3, and 4. The proposed license amendment does not affect this requirement.

The basis for the proposed TS change is based on containment integrity calculations which demonstrate that the peak containment pressure following a postulated main steamline break accident at full power will not result in a containment pressure exceeding the design pressure of 60 psig. These calculations were performed using computer codes previously found acceptable by the NRC, however, the licensee introduced several new changes to the calculation methods which have not been previously reviewed by the NRC staff.

2.0 REGULATORY EVALUATION

The NRC issued construction permits for Point Beach, Units 1 and 2, prior to February 20, 1971, the date of issuance of the final NRC rule promulgating the general design criteria (GDC) of 10 CFR Part 50, Appendix A. Therefore, 10 CFR Part 50, Appendix A, does not apply to Point Beach, Units 1 and 2.

Section 5.1.1.1, "General Design Criteria," of the Point Beach Nuclear Plant Final Safety Analysis Report discusses plant-specific GDCs. Point Beach-specific GDC 10 requires that the containment be designed to sustain large reactor coolant system pipe breaks without loss of integrity. This GDC does not, nor do the other Point Beach-specific GDCs, mention secondary system pipe breaks.

The NRC Standard Review Plan Section 6.2.1.4, "Mass and Energy Release Analysis for Postulated Secondary System Pipe Ruptures," and Section 6.2.2, "Containment Heat Removal Systems," contain guidance on the review of pressurized-water reactor (PWR) main steamline break accidents. Point Beach was also licensed prior to issuance of the Standard Review Plan. Nevertheless, the NRC staff has used the Standard Review Plan, as well as applicable Westinghouse topical reports, WCAP-8822-S2-P-A (proprietary version) and WCAP-8860-S2-A (nonproprietary version), "Mass and Energy Releases Following a Steam Line Rupture for Dry and Subatmospheric Containment Designs," dated September 1986, and WCAP-8327 (proprietary version) and WCAP 8326 (nonproprietary version), "Containment Analysis Code (COCO)," dated July 1974, as guidance in reviewing the licensee's safety analyses supporting the requested TS change.

3.0 TECHNICAL EVALUATION

3.1 Overview

The licensee's analysis postulates the rupture of a main steamline. Rupturing the main steamline results in the discharge into the containment of the contents of the faulted steam generator, as well as some of the steam in the steamlines and steam from flashing of the feedwater in the feedwater lines. This produces a rapid pressurization of the containment and a rapid increase in the temperature of the containment atmosphere. The discharged steam is condensed on the colder structures within the containment, including the containment liner. In addition to this natural process, safety systems are available to mitigate the increase in pressure and temperature. These include the containment air recirculation cooling system and the containment spray system.

The results of the licensee's reanalysis of this event are a peak containment pressure of 59.8 psig and a maximum containment temperature of 285 °F, both occurring 276 seconds after instantaneous rupture of the main steamline. These values are below the containment design pressure of 60 psig and the containment design temperature of 286 °F.

The effect of the failure of the feedwater regulating valve to close during a main steamline break accident was considered in a previous core analysis (LER 88-008-00, "Steam Line Break with Continued Feedwater Addition," dated August 12, 1988).

3.2 Mass and Energy Release into the Containment

The first step in calculating the containment pressure and temperature following a main steamline break is to determine the amount of mass and energy discharged from the faulted steam generator and the steam and feedwater lines into the containment. The licensee's analyses include conservative assumptions. These are listed in Section 4.2 of the licensee's January 11, 2002, application. Among these are assuming a power level of 102 percent of rated thermal power, increasing the reactor coolant system average temperature by accounting for uncertainties, and assuming end-of-cycle conditions with the most reactive control rod stuck out of the core. Offsite power is assumed to be available since the reactor coolant pumps then continue to operate. This increases the heat transfer from the reactor coolant to steam generator. The initial level in the steam generator is assumed to be its highest expected value (64 percent plus uncertainty). This maximizes the mass being discharged from the faulted steam generator.

The critical mass flow rate is calculated using the Moody critical flow equation. This equation is conservative for this application.

The increased flow of feedwater due to the depressurization of the faulted steam generator results in opening of the feedwater regulating valve in response to the steam flow. Flashing of the water then occurs in the unisolable portion of the feedewater line between the faulted steam generator and the main feedwater pump discharge valves. Account is taken of the difference in temperature between the unheated water upstream of the feedwater heaters (to the main feedwater pump discharge valves which are closed on a safety injection signal) and the heated water downstream of the feedwater heaters. This is done because the post-accident power generation is sensitive to the reactor coolant temperature. Maximum flowrates of the auxiliary feedwater pumps are assumed with no delay in time to injection. The auxiliary feedwater flow is assumed terminated at 600 seconds by the operator.

The assumed single failure is failure to close the feedwater regulating valve in the steamline to the faulted steam generator. Because this is the assumed single failure, the reverse flow is taken to be the steam in the unisolable volume between the faulted steam generator and the non-return check valve; that is, failure of the non-return check valve is not assumed. Because the worst failure is failure of the feedwater regulating valve to close, full emergency core cooling system (ECCS) flow could be modeled. However, the licensee has assumed that only a single train of ECCS (with a 10 percent reduction in pump flow) injects into the reactor vessel. The licensee's September 12, 2002, letter discusses the effect of assuming only one train of ECCS. The assumption is not very significant, but it does add some margin to the calculation of return to criticality, or, as the licensee states, it provides some conservatism with respect to the shutdown margin used in the analysis. The NRC staff finds the licensee's treatment of the single failure criterion to be conservative and acceptable.

Two aspects of the mass and energy analysis became the focus of the NRC staff's review:

1. The mass and energy release from the faulted steam generator into the containment considered the feedback effect of the resultant increase in containment pressure. Traditionally, even though it is not realistic, the containment pressure is conservatively assumed to remain at the maximum allowable containment pressure during normal operation prior to the accident (as specified in LCO 3.6.4).

2. The licensee included the effect of liquid entrainment in the effluent released from the faulted steam generator but did not use calculations specific to the Point Beach steam generators (Models 44F and D47).

The licensee used the Westinghouse LOFTRAN computer code (Reference 4 of the licensee's January 11, 2002, application) to calculate the mass and energy release into the containment as a result of a postulated rupture of a main steamline. The LOFTRAN computer code has been previously approved by the NRC for main steamline break analyses (WCAP-8822-S2-P-A).

Blowdown of a steam generator results in the discharge of steam. For a short period of time at the beginning of blowdown, the steam contains entrained water which lowers the enthalpy (energy content) of the discharging steam. Westinghouse has addressed this topic in several topical reports. In particular, WCAP-8822-S2-P-A discusses the calculation of water entrainment in the steam blowdown.

Entrainment of water in the blowdown discharge is a result of the "swell" of the steam generator two-phase mixture and flow reversal through the steam separator drains of the steam generator due to the sudden depressurization.

The licensee's January 11, 2002, application points out that entrainment was considered but no specific calculations were done for the Point Beach steam generator design. In its September 12, 2002, supplemental letter, the licensee justifies this approach by demonstrating that, for large breaks, the entrainment fraction and timing are not sensitive to the steam generator design. The licensee points out that WCAP-8822-S2-P-A shows that for a double-ended steamline rupture, the calculated entrainment fraction in the blowdown flow is similar for different steam generator designs. In its September 12, 2002, supplemental letter, the licensee states that "for the Point Beach analysis, a table of break effluent quality versus time was generated that conservatively bounds the data of WCAP-8822." In addition, an uncertainty of 0.1 is added to the quality of the discharging steam.

The NRC staff agrees that the calculations referenced by the licensee demonstrate that water entrainment is insensitive to the steam generator design for large breaks and the licensee's use of bounding values for entrainment based on the calculations of WCAP-8822-S2-P-A is acceptable.

Mass and energy release calculations typically assume that the containment pressure remains at the initial value (taken as the TS maximum allowable value) rather than increasing as a result of the mass and energy discharged into the containment. Although this is not realistic, it is conservative for the portion of the blowdown below critical flow conditions when backpressure must be taken into account. Including the increase in containment pressure in the mass and energy release calculations results in a transition to noncritical flow earlier in the transient. This results in a reduction in the break flow rate and hence, a reduction in the rate of increase of containment pressure. For the Point Beach analyses, the increased containment pressure is modeled in the LOFTRAN code as a function of time. The values of containment pressure as a function of time (obtained from COCO, the Westinghouse containment code) are manually input into LOFTRAN. The licensee stated that the final containment pressure input array was compared with COCO output to confirm that it is properly bounded by the COCO result.

Because this is a real effect and the calculations are done so that the final COCO result for containment pressure bounds the LOFTRAN input array, and because COCO is acceptable for licensing calculations, the NRC staff finds this approach acceptable.

For the reasons cited above, the NRC staff finds the licensee’s mass and energy release calculations to be conservative and acceptable.

3.3 Containment Analysis

The containment pressure is calculated using the Westinghouse COCO code, with the mass and energy release calculations as input. The COCO code has been previously accepted by the NRC for containment analyses.

The licensee’s January 11, 2002, application lists the initial conditions for the analysis of the Point Beach main steamline break accident. Conservative values (i.e., values maximizing the peak containment pressure) were chosen for the refueling water storage tank (RWST) water temperature (a high temperature tends to make the sprays less effective), the initial containment temperature, the initial containment pressure (set at the requested TS allowable limit of 2.0 psig), the initial relative humidity (set to a minimum value to maximize the containment air mass which results in an increased containment pressure), and the net free containment volume. The conservatism in the containment volume is discussed in the licensee’s September 12, 2002, supplemental letter.

Based on the use of an acceptable computer code and conservative assumptions, the NRC staff finds the licensee’s containment analysis to be acceptable.

3.4 TS Bases

In its January 11, 2002, application, the licensee included changes to the TS Bases associated with the proposed TS changes. The NRC staff has no objection to the licensee’s proposed changes to the TS Bases.

4.0 STATE CONSULTATION

In accordance with the Commission's regulations, the Wisconsin State official was notified of the proposed issuance of the amendments. The State official had no comments.

5.0 ENVIRONMENTAL CONSIDERATION

These amendments change a requirement with respect to the installation or use of a facility component located within the restricted area as defined in 10 CFR Part 20 or change a surveillance requirement. The staff has determined that the amendments involve no significant increase in the amounts and no significant change in the types of any effluent that may be released offsite, and that there is no significant increase in individual or cumulative occupational radiation exposure. The Commission has previously published a proposed finding that these amendments involve no significant hazards consideration and there has been no public comment on such finding (67 FR 12605). Accordingly, these amendments meet the eligibility criteria for categorical exclusion set forth in 10 CFR 51.22(c)(9). Pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared in connection with the issuance of these amendments.

6.0 CONCLUSION

The Commission has concluded, based on the considerations discussed above, that (1) there is reasonable assurance that the health and safety of the public will not be endangered by operation in the proposed manner, (2) such activities will be conducted in compliance with the Commission's regulations, and (3) the issuance of the amendments will not be inimical to the common defense and security or to the health and safety of the public.

Principal Contributor: R. Lobel

Date: November 26, 2002